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doi:10.1016/j.pnucene.2005.05.026

Progress in Nuclear Energy, Vol. 47, No. 1-4, pp. 260-270, 2005

Printed in Great Britain
0149-1970/$ - see front matter

HIGH TEMPERATURE REACTOR DEVELOPMENT IN CHINA

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ABSTRACT

Since the late 1970'-s the research and development program on the high temperature gas-cooled reactor (HTR) has been carried out in China. The 10 MW High Temperature Gas-cooled Reactor-Test Module (HTR-10) reached first criticality in 2000 and was put into full power operation in 2003. Six safety demonstration tests were done on the HTR-10. The project of the HTR-10 with a gas turbine cycle is underway. The project of the HTR demonstration plant with a power of around 150 MWe (HTR-PM) is planned. In this paper the HTR development in China is briefly described.

KEYWORDS

HTR-10; HTR; Safety demonstration test; HTR-PM

1. INTRODUCTION

The research and development program on the high temperature gas-cooled reactor in China was started in the late 1970'-s and was part of the China High Technology Program in 1986-2000. The target was to build a 10 MW High Temperature Gas-cooled Reactor- Test Module at the Institute of Nuclear and New Energy Technology (INET) by 2000. In fact the HTR-10 construction started in June 1995, achieved first criticality in December 2000 and reached full power in February 2003. After that six safety demonstration tests were carried out. Meanwhile the project of the HTR-10 with a gas turbine cycle is underway, which is involved at the second phase of the China High Technology Program (2001-2005). In parallel the project of the demonstration plant with a power of around 150 MWe has been planned and is being carried out. In this paper the HTR-10 design is described, the results of safety demonstration tests are shown and the HTR-10 with a gas turbine cycle and the HTR-PM are introduced.

2. HTR-10 DESCRIPTION

The HTR-10 design (see **Fig.1**) represents the features of modular HTR design (Lohnert, 1990). The reactor core and the steam generator are housed in two separate steel pressure vessels which are arranged in a "side-by-side" way. These two vessels are connected to each other by a connecting vessel in which the hot gas duct is designed. All these steel pressure vessels are in contact with cold helium at about 250°C coming out from the circulator which sits over the steam generator tubes in the same vessel. The HTR-10 key design parameters are listed in **Table 1**. (Sun, 2000)

Table 1. The HTR-10 main design parameters

Parameters	Unit	Value
Reactor thermal power	MW	10
Active core volume	m^3	5
Average power density	MW/m^3	2
Primary helium pressure	MPa	3
Helium inlet temperature	°C	250/300
Helium outlet temperature	°C	700/900
Helium mass flow rate	kg/s	4.3/3.2
Fuel		UO_2
U-235 enrichment of fresh fuel elements	%	17
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements		27,000
Refueling mode		Multi-pass, continuously
Average discharge burnup	MWd/t	80,000

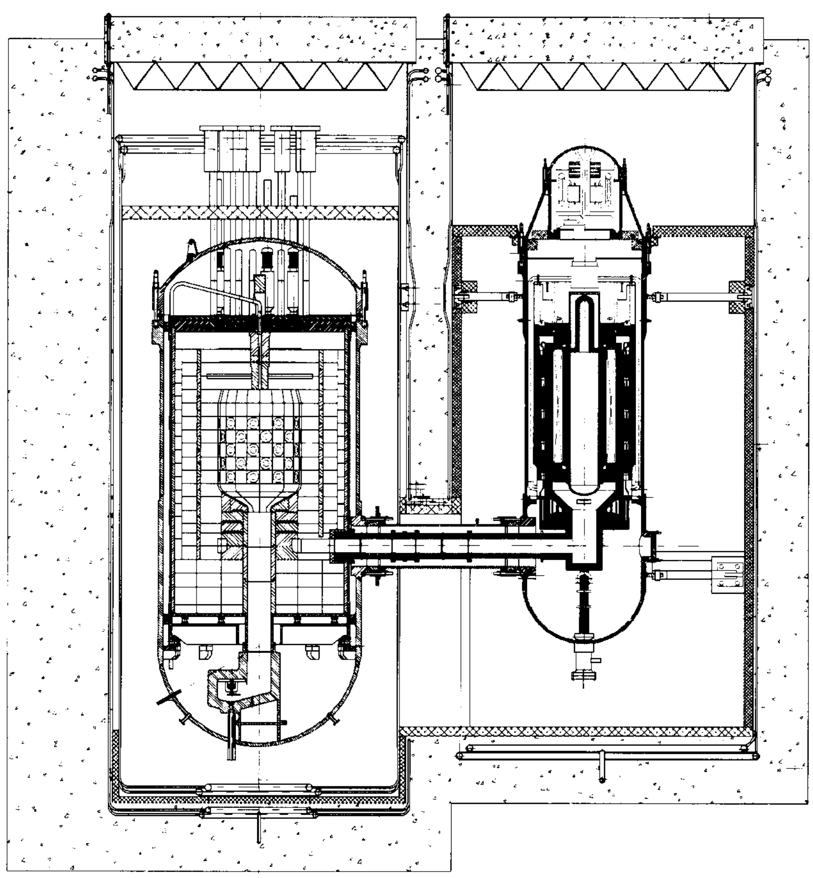


Fig. 1. Cross section of the HTR primary circuit

Fuel elements used are the spherical fuel elements (6 cm in diameter) with TRISO coated particles. The reactor equilibrium core contains about 27,000 fuel elements forming a pebble bed that is 180 cm in diameter and 197 cm in average height. The UO_2 fuel is 17% enriched. Every spherical fuel ball contains five grams of heavy metal.

Graphite serves as the main material of core structures which mainly consist of the top, bottom and side reflectors. The ceramic core structures are housed in a metallic core vessel which is supported on the steel pressure vessel. The thickness of the side reflector is 100 cm. In the side reflector, cold helium channels are designed in which helium flows upward after entering the reactor pressure vessel from between the connecting vessel and the hot gas duct.

Helium flow reverses at the top of reactor core into the pebble bed, so that a downward flow pattern takes place within it. After being heated in the pebble bed, helium enters into a hot gas chamber in the bottom reflector, and from there it flows with reactor outlet temperature through the hot gas duct to the heat exchanging components.

The steam generator is composed of a number of modular helical tubes which are arranged in a circle between two insulation barrels inside the steam generator pressure vessel. The place inside the inner barrel was originally planned for an intermediate heat exchanger (IHX) which was to be installed in the second phase of the project, but this plan has been changed.

Decay heat removal from the HTR-10 under accident conditions is designed to be completely passive. Upon a loss of pressure accident, during which no core cooling is foreseen at all, decay power will dissipate through the core structures by means of heat conduction and radiation to the outside of the reactor pressure vessel, where, on the wall of the concrete housing, a surface cooling system is designed. This system works on the principle of natural circulation of water and it conduct the decay heat via air coolers to the atmosphere. In fact, this surface cooling system is designed to protect the vessel and concrete structures more than the ceramic reactor core from being overheated by decay power.

There are two reactor shutdown systems, one control rod system and another small absorber ball system. They are both designed in the side reflector. Both systems are able to bring the reactor to cold shutdown conditions. Since the reactor has strong negative temperature coefficients and decay heat removal does not require any circulation of the helium coolant, the turn-off of the helium circulator can also shut down the reactor from power operating conditions.

The reactor confinement includes the reactor cavity, the steam generator cavity and the cavities which house the main process systems, like the primary pressure relief system, the helium purification system and the fuel handling system, which are directly connected to the primary system.

3. HTR-10 SAFETY DEMONSTRATION TESTS

The aim of safety demonstration tests is to demonstrate the inherent safety features of modular HTRs; in particular, the goal was to obtain the core and plant transient data for the validation of computer codes for safety and transient analysis to assure that the maximum fuel temperature would not exceed 1600°C and that the temperature of components, including pressure vessels and their supports, the reactor internal as well as the concrete of the cavity, would not exceed the temperature limits of the respective materials. In addition, safety demonstration tests done on a real reactor will persuade peoples to believe that the high temperature gas cooled reactor is an inherent safety reactor and will also persuade safety authorities to use new design criteria for the HTGR instead of the original ones, which are mainly based on the light water cooled reactor experience.

Six safety demonstration tests were done on the HTR-10. They are the loss of helium flow at 30% of rated power; turbine trip at full power; loss of off-site power supply at 30% of rated power; the helium circulator trip without scram at 30% of rated power; reactivity insertion (5 mk) without scram at 30% of rated power and the helium circulator trip without closing outlet cut-off valve at 30% of rated power.

To assure that no components or systems would be failure or be seriously damaged, the temperature of the surface of the reactor pressure vessel, the steam generator pressure vessel and the hot gas duct pressure vessel, the temperature of the reactor internal as well as the temperature of the shielding concrete and supports of the reactor pressure vessel were monitored during the safety demonstration tests. The temperatures limits for these were set to 350°C, 400°C and 70°C respectively. As soon as any one of the monitored temperatures was reached the temperature limit, the safety demonstration test would be stopped by the operator.

In addition to maintain the integrity of the primary circuit and to avoid the release of helium to the surrounding, the primary pressure was also monitored at all times. The pressure limit was set to 3.3 MPa (The set pressure for the safety valve of the primary circuit is 3.39 MPa.) As with the monitored temperatures, the safety demonstration test would be stopped by the operator and the helium would be release to the helium storage tank, if necessary, if the primary pressure should exceed the pressure limit.

3.1 The helium circulator trip without scram at 30% of rated power

Before the test the reactor was steadily operated at 3 MWt with an outlet temperature of 650°C, an inlet temperature of 215°C and a primary pressure of 2500 kPa. And then the helium circulator power was switched off by the operator. Meanwhile all of the control rods could not be automatically dropped down into holes due to no signal for the action of the reactor protection system even in the case of one of the protection parameters being beyond the set point. Of course other actions, including closing of the cut-off valve of the helium circulator, stopping of the main feed water pump and closing of the isolation valve at the second circuit and so on, would still be implemented through the reactor protection system when one of the protection parameters exceed a set point.

Because of the characteristics of components and systems of the HTR-10, the reactor protection system started to implement all of the actions except the dropping down of the control rods at the12th second after the helium circulator was switched off at which time the ratio of the helium flow to the water flow exceeded the set point. At the 17th second the reactor power was decreased to the level of residual heat power due to the negative reactivity feedback effect caused by the increase of the core temperature. At the 2600th second the reactor again reached criticality due to the action of a positive reactivity feedback effect caused by the decrease of the core temperature, from cooling by the surface cooling system. After several oscillation of the power, the reactor reached a steady-state power of 180 kW at around the 7200th second. The level of the steady-state power depends on the capacity of the surface cooling system and heat loss. Throughout the entire period all of the safety parameters, including the temperatures of the reactor internal, the surface of the three pressure vessels and the shielding concrete as well as the primary pressure were within the operational set limits. The transient of the power and the revolution of the helium circulator as well as the primary pressure are shown in **Figs. 2** and **3** respectively.

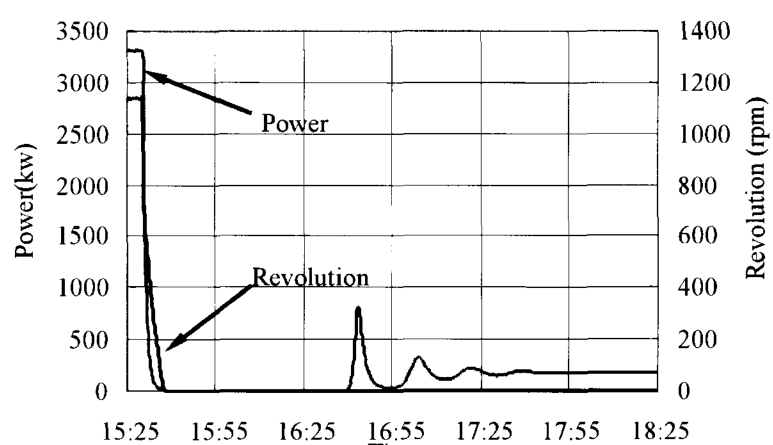


Fig. 2. Transient of the power and the revolution of the helium circulator

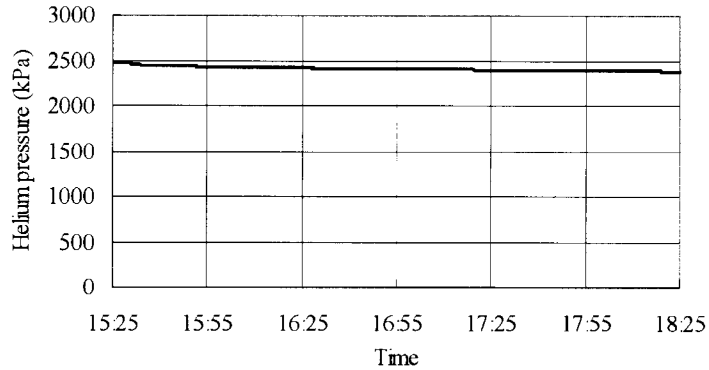


Fig. 3. Transient of the primary pressure

3.2 Reactivity insertion (5mk) without scram at 30% of rated power

Before the test the reactor was steadily operated at 3 MWt with the outlet temperature of 650°C, the inlet temperature of 215°C and the primary pressure of 2500 kPa. And then one control rod was withdrawn by the operator. Meanwhile all of the control rods could not be automatically dropped down into holes due to no signal for the action of the reactor protection system even in the case of one of the protection parameters being beyond the set point. Of course other actions, including stopping of the helium circulator, closing of the cut-off valve of the helium circulator, stopping of the main feed water pump and close of the isolation valve at the second circuit and so on, would still be implemented through the reactor protection system should one of the protection parameters exceed the set point.

Because of the characteristics of components and systems of the HTR-10, the reactor protection system started to implement all of the actions except dropping down of the control rods at the 13th second after the control rod was withdrawn, at which time the rate of power increase exceeded the set point. At the 15th second the reactor power reached 1.2 times the initial operational power. At the 30th second the power reached the highest value (around 2.4 times the initial operational power). At the 700th second the reactor power was decreased to the level of residual heat power due to the negative reactivity feedback effect caused by the increase of the core temperature. At the 2520th second the reactor again reached criticality due to the positive reactivity feedback effect caused by the decrease of the core temperature from cooling by the surface cooling system. After several oscillation of the power, the reactor reached a steady-state power of 180 kW at around 7200th second. The level of the steady-state power depends on the capacity of the surface cooling system and heat loss. Throughout the entire period all of safety parameters, including the temperatures of the reactor internal, the surface of three pressure vessels and the shielding concrete as well as the primary pressure were within the operational limits. **Figs. 4** and **5** show the primary pressure as well as the transient of the power and the position of the control rod, respectively.

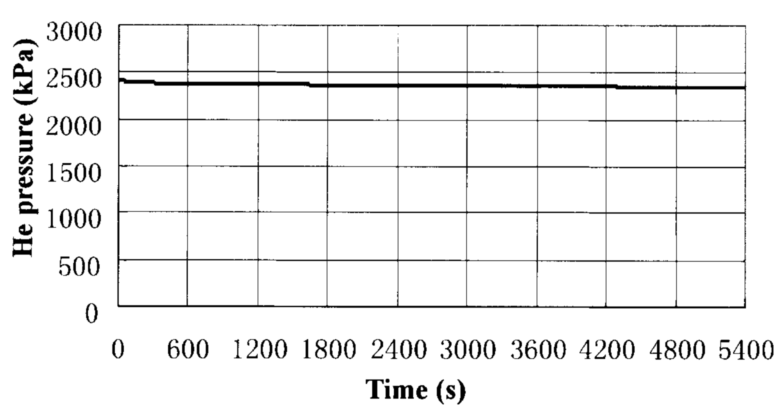


Fig. 4. Transient of the primary pressure

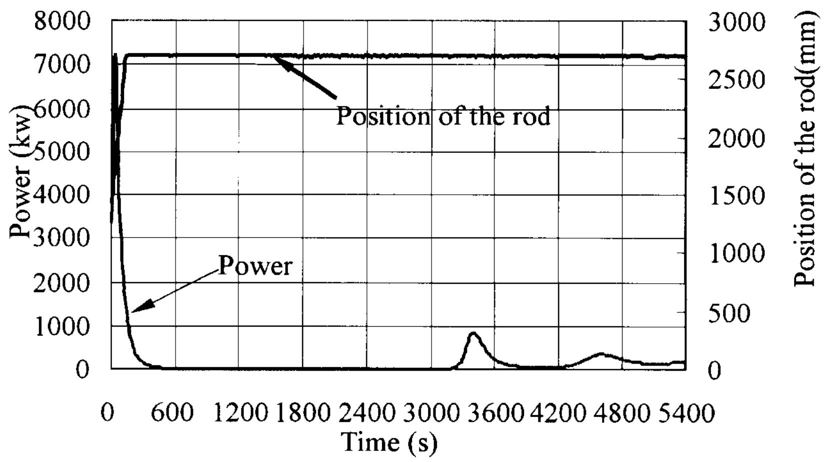


Fig. 5. Transient of the power and the position of the control rod

The results of safety demonstration tests show that the HTR-10 has inherent safety features and computer codes for the design of the HTR-10 are validated and can be used for modular HTR design.

4. HTR-10 WITH A GAS TURBINE CYCLE

Since the late 90s the version of the HTR with a gas turbine cycle has been considered by many research institutes and companies throughout the world (Barnert and Kugeler, 1995), based on the achievements in the technology of gas turbines, high-performance heat exchangers and electromagnetic bearings. The HTR with gas turbine cycle can reach higher electric efficiency (up to 50%) and simplify the power unit design so it therefore can reduce both capital and operational cost.

However, there is no any practical experience in the use of the HTR with a gas-turbine cycle. In order to get more operation experience, the project of the HTR-10 with gas turbine cycle was set up at the second phase of the China High Technology Program in 2001-2005. The INET and the OKBM (Russia) signed a contract to start the conceptual design of the power conversion unit (PCU) with a direct gas-turbine cycle for the HTR-10 at the end of 2000. It was jointly completed in 2002. The basic design was completed by the INET in 2003. The main components are under manufacture and it is planned to be put into operation in 2006.

The conceptual design for the PCU adopted recently technical achievements with regard to gas turbines, electromagnetic bearings, heat-resistance materials, and effective heat exchangers. (Lei, 2003) The design of the PCU has the following technical features: direct closed gas-turbine cycle was used; all of the components are installed within the original steam generator pressure vessel; the turbine, compressors and the generator are mounted on a shaft; the turbomachine shaft is supported by electromagnetic bearings; the precooler and the intercooler adopt modular design and the generator and the turbocompressor are located in different unit cavities with a seal between them, limiting coolant leakage from the turbocompressor cavity to the generator cavity.

The flow diagram and parameters of the HTR-10 with a gas turbine cycle were selected based on the following requirements: reasonable efficiency of about 22% was chosen to ensure reliable validation of analytical parameters and acquisition of data for the verification of computer codes; limitation of primary pressure and the temperature at the reactor inlet are 3.0 MPa and 330°C; allowable relative bypass flow is 3% and allowable relative pressure loss is less than 10%. The flow diagram and main parameters are shown in **Fig. 6** and **Table 2** respectively.

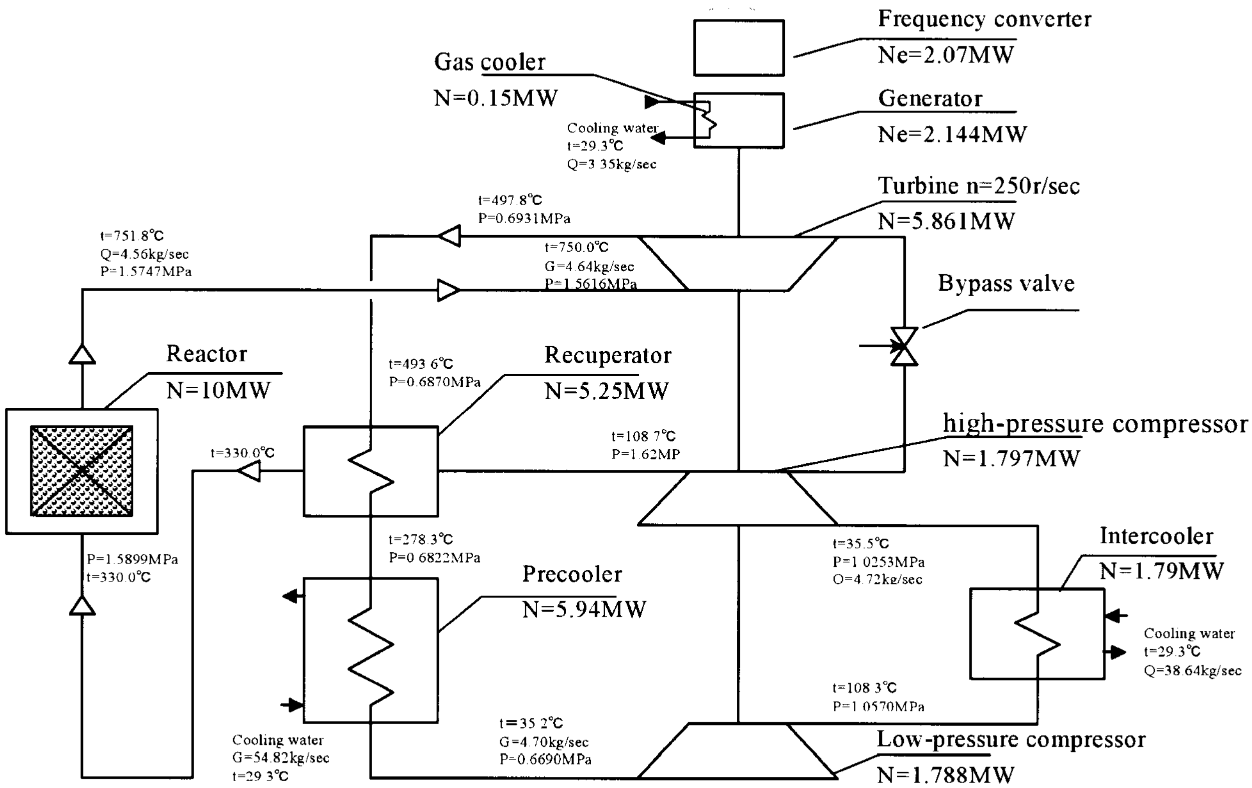


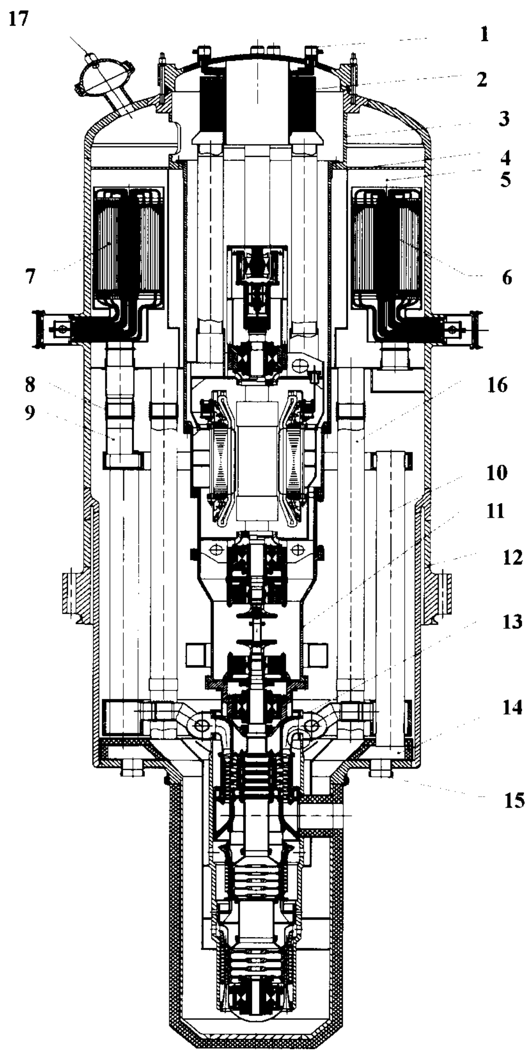
Fig. 6. Thermal circuit of HTR-10 with gas-turbine cycle

Table 2. Main parameters of HTR-10 with direct gas-turbine cycle

Parameter	Value
Reactor	
Reactor power, MW	10.00
Temperature at the core inlet/outlet, °C	330/751.8
Pressure at the inlet/outlet, MPa	1.5899/1.5747
Helium flowrate, kg/s	4.56
Power conversion unit (PCU)	
Thermal circulation efficiency, %	22.765
Generate electricity efficiency, %	20.694
Total relative pressure loss, %	9.7467
Total relative helium leaks, %	5.3
PCU mass, t	51.9
PCU height, mm	7940
Water temperature at the PCU inlet, °C	29.3

Helium heated up in the reactor core goes along the hot gas duct into the turbine inlet. After expansion in the turbine, helium flows along the recuperator low-pressure side, transferring heat to helium counter-flowing along the recuperator high-pressure side. Further helium cooling takes place in the precooler. Then helium is compressed in the low-pressure and high-pressure compressors with intermediate cooling in the intercooler. After flowing through the high-pressure compressor, helium flows along the recuperator high-pressure side, where it is heated up due to heat exchange with helium coming out of the turbine. Heat removal from the PCU cooler is performed by the intermediate cooling water circuit. Thus heat transferred from the reactor to the PCU is successively used in the gas-turbine and steam-turbine cycles.

The arrangement of the PCU is shown in **Fig.** 7, which includes the following components: PCU vessel; turbomachine consisting of the turbocompressor, the generator and connecting coupling; recuperator, precooler, intercooler and generator gas cooler; in-vessel metalworks.



1 nozzle
2 gas cooler
3 shell
4 plate
5 chamber
6 intercooler module
7 precooler module
8 expansion pieces
9 pipeline
10 recuperator
11 turbomachine
12 PCU vessel
13 header
14 header
15 nozzle
16 pipeline
17 lead-out

Fig. 7. PCU structural diagram

5. HTR-PM PROJECT

The successful operation of the HTR-10 and the excellent results of safety demonstration tests both give a preliminary demonstration that the design of the HTR-10 is successful and that the HTR-10 has inherent safety features. The HTR will be one of the advanced reactor candidates for the next reactor generation if the economics for the HTR plant compare favorably both with those for other types of reactors and with those for conventional power plants. Therefore the plan of building a HTR demonstration plant was proposed by China Huaneng Group (CHNG), China Nuclear Engineering and Construction Corporation (CNECC) and Tsinghua University (INET).

The CHNG is the largest utility in China. This utility has great experiences in building conventional power plants and in building many conventional power plants with various power rates, which were the first kind of the conventional power plant. The CNECC is the only company to install components and systems of the nuclear island in China. The company has great experiences in the installation of various reactors, including the pressurized water reactors and the HTR-10. The INET was the institute for the design of the HTR-10.

Three parties signed the MOU to build a HTR demonstration plant-pebble bed modular (HTR-PM) on 1st March 2004. After that a preparation team for setting up the HTR-PM project was established, including staffs from three parties. The pre-feasibility of the HTR-PM project is under way. The selection of the site for the HTR-PM project was carried out. There are two candidates for the site of the HTR-PM project. One is in Shandong Province and the other is in Anhua Province. By the end of this year the pre-feasibility study is expected to be completed and then a project proposal will be submitted to the National Development and Reform Committee (NDRC) to get approval to set up the HTR-PM project.

The design of the HTR-PM will be similar to the design of the HTR-10, except they have rather different power rate. In addition in order to increase the power rate the design of annual core will be chosen instead of one of single zone and at second circuit the scheme of the re-heat was adopted to have higher power efficiency. These measures can reduce both the unit capital investment and operational cost to improve the economic competitiveness.

In the HTR-PM design, it is intended to reflect as much as possible the past experience and lessons from worldwide HTGR development, and to use the proven methodologies and technologies of the HTR-10 test reactor. Now that there is the real project background, the mature steam turbine cycle has been chosen for power generation in order to avoid too much R&D and to shorten the overall time period of the demonstration project. (Sun, 2004)

The HTR-PM design has the following striking technical features:

- Spherical fuel elements with TRISO coated particles are used, which have proven capability of fission product retention up to 1,600°C in case of accident.
- A two-zone core design is adopted, with one central movable column of graphite spheres surrounded by pebble fuel elements. The purpose of the two-zone core design is to increase the power output of one reactor module while maintaining the passive decay heat removal features.
- The active reactor core shall be surrounded by ceramic materials of graphite and carbon bricks, which are high-temperature resistant.
- Decay heat in the fuel elements shall dissipate by means of heat conduction and radiation to the outside of the reactor pressure vessel, and then be taken away to the ultimate heat sink by water cooling panels on the surface of the primary concrete cell. Therefore, no coolant flow through the reactor core shall be necessary for decay heat removal in case of loss of coolant flow or loss of pressure accidents. Maximum accident fuel temperature shall be limited to 1,600°C.
- Spherical fuel elements shall be charged and discharged continuously in a so-called "multi-pass" mode, which means that before the fuel elements reach the discharge burn-up, they will go through the reactor core several times.
- Two independent reactor shutdown systems are foreseen. Both systems shall be designed in the side reflector graphite blocks. The neutron absorber elements shall fall into the designated channels in side reflectors by gravity when called upon.
- The reactor core and the steam generator are housed in two steel pressure vessels which are connected by a connecting vessel. Inside the connecting vessel, the hot gas duct is designed. All the pressure-retaining components, which comprise the primary pressure boundary, are in contact with the cold helium at the reactor inlet temperature.
- At a complete loss of pressure accident, the primary helium inventory shall be allowed to be released into the atmosphere. Then the helium release channel shall be closed and the reactor building shall be vented and serves as the last barrier to radioactivity release.
- Several of HTR-PM modular reactors can be built at one site to satisfy the power capacity demand of the utility. Some auxiliary systems and facilities shall be shared among the modules.

Figure 8 shows the cross section of the HTR-PM and its main parameters are listed in **Table 3.**

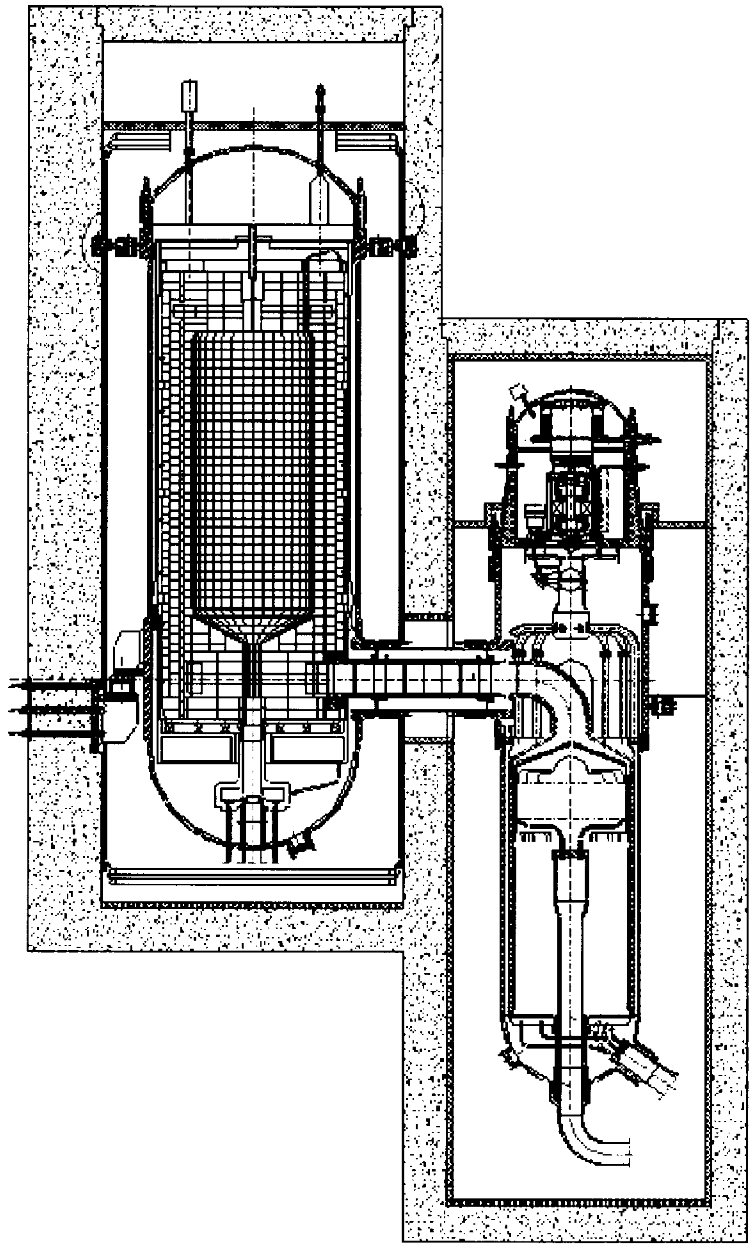


Fig. 8. The cross section of the HTR-PM

The core inlet temperature (250°C) is the same as that of the HTR-10 and the core outlet temperature (750°C) is higher than that of the HTR-10 to match the steam temperature at the second circuit to reduce the volume of the steam generator. The higher primary pressure is chosen to have higher heat transmission capability. The flow scheme is the same as the one of the HTR-10. Fuel elements are the same as that of the HTR-10 except for the uranium enrichment and the number of coated particles in the ball. Of course the parameters which are determined by the power rate, such as the number of the fuel elements, helium flow rate and so on, are quite different from those of the HTR-10.

It is planned that the HTR-PM construction will start at the beginning of 2007 and that first criticality will be achieved by the end of 2010.

Table 3. Main parameters of the HTR-PM

Reactor thermal power	MW	450
Active core diameter/height	m	2.20-4.00/11.0
Average power density	MW/m^3	4.67
Primary helium pressure	MPa	7.0
Helium inlet temperature	°C	250
Helium outlet temperature	°C	750
Helium mass flow rate	Kg/s	172
Fuel		UO2
U-235 enrichment of fresh fuel elements	%	9.08
Diameter of spherical fuel elements	mm	60
Number of spherical fuel elements	ball	520000
Number of graphite balls	ball	225530
Average discharge burnup	MWd/tU	80,000
Refueling mode		Multi-pass continuously
Number of control rods	set	18
Number of small absorb ball systems	set	18
Main steam pressure	MPa	13.45
Main steam temperature	°C	543
Main steam flow rate	t/h	557.3
Feed water temperature	°C	205.3
Power from steam turbine	MW	200

6. CONCLUSION

The HTR research and development program in 1986-2000 was well implemented. The HTR-10 achieved first criticality in 2000 and reached full power in 2003. The operation of the HTR-10 was quite good. The results of the safety demonstration test show that the HTR-10 has inherent safety features and the computer codes for the HTR-10 are validated and can be used for the design of the module HTR. The basic design of the HTR-10 with a gas turbine cycle was carried out. It is planed to be put into operation in 2006. The preparation of the HTR-PM project was started. The set up of the HTR-PM project will be approved by end of this year and this project is planed to be finished in 2010.

ACKNOWLEDGMENT

The authors sincerely thank all of the staffs participating in the design, construction and operation of the HTR-10 and those joining the recent two projects for their efforts to realize our target at short period. The authors sincerely thank the national authorities for supporting us in the completion of these projects.

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